FENOC
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
10 Center Road
Perry, Ohio 44081

John K. Wood
Vice President, Nuclear

440-280-5224
Fax: 440-280-8029

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United States Nuclear Regulatory Commission
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Perry Nuclear Power Plant
Docket No. 50-440

Ladies and Gentlemen:

Enclosed is Licensee Event Report 2000-001, "Potential for Inadequate Suppression Pool Make-Up for the Emergency Core Cooling Systems."

If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K Wood

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

IE22

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4) Potential for Inadequate Suppression Pool Make-Up for the Emergency Core Cooling Systems

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
1	13	2000	2000	-- 001 --	00	2	14	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	088		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Sterling W. Sanford, Senior Compliance Engineer	(440) 280-5361

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURE	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

During an engineering review of the Inclined Fuel Transfer System (IFTS) blind flange removal at power, the Emergency Core Cooling System (ECCS) was determined not to meet the design criteria due to a potential for insufficient Suppression Pool Make-Up (SPMU). A recent license amendment allowed removal of the blind flange and subsequent testing of IFTS at power. The blind flange is part of the upper pool boundary for water volume required for the Loss of Coolant Accident (LOCA) analysis.

The boundary to maintain upper pool volume, with the blind flange and IFTS gate in the upper containment pool removed, is either the upper or bottom IFTS valve. These valves are not safety related. The safety analysis report credits only safety-related equipment for accident mitigation, and, therefore, these valves are not assumed to prevent water loss during a LOCA. The reduced water inventory would be less than that required for SPMU. The resultant reduced suppression pool volume and increased temperature could potentially result in the loss of all ECCS pumps due to a lack of adequate suction pressure.

On March 25, 1999, the IFTS blind flange and the IFTS pool gate were removed at power prior to achieving cold shutdown on March 27, 1999, for refueling outage seven. This condition was reported in accordance with 10 CFR 50.72 (b)(1)(ii)(B) for the plant having been outside of the design basis on January 13, 2000 (ENF # 36588). This Licensee Event Report is submitted in accordance with 10 CFR 50.73 (a)(2)(ii) for the plant having been outside of the design basis.

The cause of this condition was attributed to inadequate internal reviews and insufficient procedural barriers.

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TEXT *(If more, use space is required additional copies of NRC Form 366A)* (17)

I. INTRODUCTION

The Inclined Fuel Transfer System (IFTS) [DF] is used during refueling operations and functions to transfer fuel between the reactor and the spent fuel pools. An operating license amendment was recently obtained to allow for testing of the IFTS during Modes 1, 2, and 3. A blind flange must be removed from the IFTS transfer tube to allow testing between the upper containment pools and the lower elevation spent fuel pools. The blind flange and the IFTS pool gate are safety-related components and either provides part of the upper containment pool boundary to ensure that a sufficient water volume exists for Suppression Pool Make-Up (SPMU) [BT] as credited in the Loss of Coolant Accident (LOCA) analysis. The SPMU function transfers water from the upper containment pool to the suppression pool after a LOCA. During a LOCA, a large volume of water can be held up in various locations, such as the drywell and the reactor vessel, which reduces suppression pool water level. The additional makeup water from the upper containment pool is used as part of the long-term suppression pool heat sink. The potential for loss of upper pool water volume with the IFTS blind flange and gate removed had not been addressed.

II. DESCRIPTION

Due to questions regarding adequacy of the license amendment, the Independent Safety Engineering Group (ISEG) was conducting a team review of Primary Containment capability with the IFTS blind flange removed. On January 13, 2000, the Design Engineering team member determined that a potential for a loss of all Emergency Core Cooling Systems (ECCS) existed during certain accident conditions. With the IFTS flange and upper containment pool IFTS gate removed, the boundary to maintain upper containment pool water volume is either the IFTS bottom valve or the upper most IFTS valve [ISV] assembly. The upper assembly is commonly referred to as the flap valve (due to the valve resembling a flap) and contains a cable sheave box and other miscellaneous piping.

The safety analysis report requires that the design of ECCS is such that failures of interfacing systems shall not affect the performance of ECCS. The safety analysis report credits only safety-related equipment for accident mitigation, and, therefore, these valves are not assumed to prevent water loss during a LOCA. With the IFTS pool gate removed, the water loss could potentially reduce the upper pool water inventory to less than that required for SPMU. The subsequent reduced suppression pool volume and the increased suppression pool temperature could potentially result in the loss of all ECCS pumps due to a loss of adequate suction pressure. Therefore, the ECCS would not meet the criteria defined in the design basis.

At approximately 0330 hours on March 25, 1999, the blind flange and IFTS pool gate were removed to allow for IFTS testing. The plant was at approximately 88 percent thermal power (power coast down) prior to entering the seventh refueling outage. The plant achieved cold shutdown about noon on March 27, 1999, and the SPMU function was no longer required. During this period of approximately 57 hours, the IFTS valves were relied upon as the upper containment pool boundary with the IFTS pool gate removed. No other systems, structures, or components were inoperable that would have contributed to this outside of design basis condition.

At 1704 (EST) hours on January 13, 2000, a notification was made in accordance with 10 CFR 50.72(b)(1)(ii)(B) for the plant having been in a condition that was outside of the design basis (ENF 36588).

III. CAUSE

Two primary causes were identified that contributed to this condition. The causes were inadequate internal reviews and insufficient procedural barriers. This license change was a previously approved amendment at another facility. The scope and reviews of the evaluations performed were inappropriately limited to the content of that amendment. Additionally, plant procedures did not specifically require multi-disciplined interface reviews for non-modification changes to the design or licensing basis.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

IV. SAFETY ANALYSIS

Engineering judgement is that the IFTS valves would have been expected to remain intact following a LOCA and a safe shutdown earthquake. This judgement is based on the known performance of the valves during the refuel outage. During the time that the flange was removed at power, the valves did not exhibit leakage in excess of normal make-up to the upper containment pools. The bottom valve is non-safety but is seismically qualified.

The loads on the closed flap valve are not significantly increased during a LOCA. The containment pressure is transmitted to both sides of the valve due to the transfer tube vent lines and cable guides and, therefore, the pressure on the valve is predominantly due to the head of water. Additionally, seismic loading of the flap valve is considered to be minor when compared to the water head pressure loading. The loads on the seismically qualified bottom valve would be equivalent to water head plus LOCA pressure, which would be less than the valve rating.

The postulated safety consequences were minimal. The probability of a LOCA and a concurrent failure of these valves during the approximately 57-hour period was extremely remote (on the order of E-10). The individual plant examination (IPE) would require no further mitigative actions for an event of this risk frequency. Additionally, Regulatory Guide (RG) 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, would classify this condition as a very small risk. The combined event probability is so low that this is not considered to be credible.

V. CORRECTIVE ACTIONS

At the time of discovery, the IFTS blind flange was already administratively controlled to prevent removal at power due to issues raised from the same ISEG team assessment. The flange will remain installed at power until all issues are resolved.

The following items are being tracked through the corrective action program by condition report 99-3035.

The engineering support personnel, the Plant Operations Review Committee, and the Company Nuclear Review Board are to review the lessons learned for this condition.

Several follow-on actions are to be performed. The first will resolve this IFTS issue. An adequacy review of recently approved license amendments is in progress. Additionally, the design interface evaluations for recent licensing and design basis changes are to be reviewed.

The procedures governing the modification and the license amendment processes are to be revised to include the requirement to evaluate for multi-discipline reviews for non-modification changes to the design and licensing basis.

VI. PREVIOUS SIMILAR EVENTS

A review of Perry Nuclear Power Plant (PNPP) Licensee Event Reports (LER) from the last 5 years was performed. There were no corrective actions associated with any LERs for the plant being in a condition outside of design basis or LERs of similar causes that would have reasonably been expected to have prevented this condition.

Energy Industry Identification System (EIIS) Codes are identified in the text by square brackets [XX].